NON-PUBLIC?: N
ACCESSION #: 9212010278
LICENSEE EVENT REPORT (LER)

FACILITY NAME: BIG ROCK POINT PLANT PAGE: 1 OF 08

DOCKET NUMBER: 05000155

TITLE: LOSS OF TRANSMISSION SYSTEM RESULTING IN MANUAL REACTOR SCRAM (RPS
ACTUATION) AND ELECTRIC AND DIESEL FIRE PUMP START (ESF ACTUATION)
EVENT DATE: 10/29/92 LER #: 92-014-00 REPORT DATE: 11/25/92

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 098

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Michael D Bourassa, Senior TELEPHONE: (616) 547-6537
Technologist

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: FK COMPONENT: 52 MANUFACTURER: S375
B VA CLR X999
B IL MON G080
REPORTABLE NPRDS: N
N
N

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On October 29, 1992 at 1111, the facility experienced a momentary interuption of station power. The main transformer protective relay operated, tripping the 138 kV tie breaker and the generator output breaker. Station Power was automatically transfered from the 138 kV line to the 46 kV line. Consequently the turbine was automatically tripped. The reactor was also manually scrammed at 1112. An Unusual Event (UE) was conservatively declared at 1135. The UE was exited at 1426.

Upon investigation, a loose connection was discovered in the current transformer terminal block of the 138 kV tie breaker. This resulted in thermal damage and separation of one of the inputs to the protective relays causing a false differential condition.

The terminal block was repaired and tested. The normal lineup for station power was restored on October 30, 1992. Containment pipeway coolers that exhibited leakage during the event were repaired and returned to service. Following an evaluation of a high differential temperature condition on the reactor vessel, the Plant was returned to power operation on November 2, 1992.

END OF ABSTRACT

TEXT PAGE 2 OF 08

DISCUSSION OF EVENT

On October 29, 1992, the reactor (RCT) was operating near 74 MWe (about 98% power). Operations were routine. The No. 1 Control Operator (CO1) was in the control room (NA), the No. 2 Control Operator (CO2) was at the turbine (TUR) performing a routine shift inspection and the Shift Supervisor (SS) was in his office. An electrical contractor was working in the plant substation (FK).

Time Event

1111 The Main Transformer (XMFR) 387 differential protective relay (87) operated, resulting in the following actions:

a. Automatic transfer of station power from the 138 kV transmission line (normal source for offsite power) to the 46 kV transmission line.

b. Generator (TG) output breaker (BKR) (116 OCB) tripped open isolating the generator.

c. The 286 lock-out relay (86) energized the hand trip solenoid (SOL) (HTS), which successfully tripped the turbine by initiating turbine stop valve (FCV) closure. Concurrent with this sequence of events, the CO2 heard turbine speed increase and observed a speed of 3993 rpm on the digital turbine rpm gauge. This speed was sufficient to initiate an overspeed trip and the overspeed mechanism did actuate. However, by the time the turbine overspeed setpoint was reached, stop valve closure had already been initiated by the HTS.

1112 Overhead lights (LF) went out momentarily in the control room. The CO1 observing that the 138 kV tie breaker (BKR) (199 OCB) (52), 116 OCB (52) and turbine were tripped, manually scrammed the reactor. The reactor scram was successful and power decayed to expected post-trip levels.

Note: When the operators were verifying full control rod (AA) insertion, three drives (D3,D6 and F4) suffered a temporary loss of position indication (ZI) because of overtravel. In accordance with System Operating Procedure 38, the safety system (JE) was reset and position indication returned. The control rods were verified fully inserted at 1126 hrs. This condition was noted during two previous scrams, (LER 92-009 and 010).

The reactor recirculation and feedwater pumps (AD;SJ;P) tripped off when the 199 OCB opened. The momentary interruption in power during the transfer to the 46 kV line caused the 2400 Vac recirculation and feed pump breakers (AD;SJ;BKR) to open on undervoltage by design.

The lower voltage pump breakers (e.g., service water and reactor cooling water pumps) (KE;CC;P) did not open and the pumps restarted when power was established from the 46 kV line.

The Emergency Diesel Generator (EK) (EDG) automatically started on loss of load on the 2B bus (BU) per design. Its output breaker did not close in because of the successful transfer to the 46 Kv transmission line.

1112 Entered 3-day Limiting Condition of Operation in accordance with Technical Specification 11.3.5.3.A.8, loss of the 138 kV transmission line. (This LCO was exited on October 30, 1992 at 2010 when the 138 kV transmission line was returned to service.)

1113 Collapsing voids in the core combined with the lack of feed flow caused steam drum (SD) level to drop to minus 20 inches below centerline, 3 inches below the Emergency Operations Procedure (EOP) entry level of minus 17 inches. The Electric and Diesel Fire Pumps (JE) started automatically and were confirmed available for core spray (BM) (see discussion below). The SS entered the level control portion of the EOPs and directed the start of the No. 1 Reactor Feed Pump (SJ;P) to restore steam drum level.

Discussion

The low steam drum level signal initiates a two-minute delay which allows a containment (NH) evacuation interval prior to system blowdown and also allows operator input to the system initiation logic. The low steam drum level signal is also used to generate a fire pump start signal. A source of core spray water is made available at the core spray valves when the fire pump discharge pressure is at or exceeding 100 psig. If a low reactor water level signal is generated, and with the two other logic inputs present (time delayed low steam drum level and core spray water availability) an automatic trip will occur and the Reactor Depressurization System will be activated. However, the low reactor water level input was never received, and the system did not actuate.

In accordance with guidance received from the Office for Analysis and Evaluation of Operational Data (AEOD) in a letter dated August 29, 1991, even though the logic was not completed this event is reportable because the fire pumps are considered part of the engineered safety features system, and the actuation of the fire pumps occurred as a result of a valid actuation signal (i.e., steam drum water level less than minus 17 inches).

1131 The No. 1 Reactor Feed Pump was secured. The Steam Drum level had been brought up to plus 20 inches above the centerline per the EOPs.

1135 The SS conservatively elected to declare an Unusual Event due to the unusual set of circumstances being experienced.

1137 The No. 1 Reactor Recirculation Pump Discharge Valve (AD;P;FCV) was closed in preparation for restarting the No. 1 Reactor Recirculation Pump. During the time it took for the valve to close, the steam drum level increased and pegged high (plus 25 inches). To reduce drum level before starting the Recirculating Pump, the Cleanup System (CE) was placed in service. This action was performed to ensure that a pressure surge would not be created that could possibly lift a primary Safety Relief Valve (RV), or inadvertently cause a slug of water to the turbine via the open Main Steam Isolation Valve (TA;FCV) to enter.

1148 Reactor Pressure increased from 800 psig logged at 1136 to 837 psig.

1156 The Spent Fuel Pool Area Monitor (DA;MON) alarmed. Areas 1 through 5 were "lost" due to a blown fuse (FU). By 1229, Health Physics had surveyed the areas and verified them to be below the area monitor setpoints.

1156 The Cleanup System was lined up for direct blowdown.

1214 The Reactor Vent, MO-N004 (RPV;VTV) was opened to insure that a steam pocket would not form in the reactor head while the Recirculating Pumps were off.

1217 Reactor pressure steadily increased from 837 (logged at 1148) to 862 psig.

1218 The No. 1 Reactor Recirculating Pump was started. Reactor pressure decreased to 854 psig.

1221 The blowdown through the Cleanup System was terminated.

1239 The No. 1 Reactor Feed Pump was restarted.

1300 A Continuous Air Monitoring (MON) alarm was received in Containment. The Enclosure Dirty Sumps' (NH;WK) level began to increase. An investigation was initiated to determine cause of level rise.

1312 The EDG was secured.

1320 Containment Pipeway Cooler (NH;CLR) Service Water (KE) leak was discovered during a tour of the Recirculating Pump Room.

1347 The Pipeway Cooler Service Water Isolation Valves (CLR;KE;ISV) were closed, and by 1400 the Enclosure Dirty Sump levels returned to normal.

1426 Exited Unusual Event.

As a result of the recirculating pumps being idle while the control rod drive pumps (AA;P) were operating, the procedural guidelines not to exceed 150 degrees F temperature differential between any two points on the reactor vessel was exceeded. Thermocouple points 18, 19 and 20 are located at the bottom of the reactor vessel (RPV) at an elevation below the bottom of the coolant inlet nozzles (NZL). Points 18 and 20 showed a linear decrease in temperature over a one hour period of approximately 215 degrees F (averaged). Point 19 decreased approximately 75 degrees F over the same period. All other thermocouple points on the reactor vessel and steam drum were normal, that is they decreased slowly, well within the 150 degree F requirements.

CAUSE OF THE EVENT

Root Causes

1. States (S375) Terminal Block/Connector (BLK;CON)

Connections on the current transformer (XCT) for the 199 OCB were found to have thermal damage that resulted in a separation of one of the required inputs (Y-phase) to the 387 protective relay. Failure of the termination point has been attributed to a loose connection. The loose connection is believed to be the result of cyclic thermal stresses over time (caused by temperature difference during periods of plant shutdown and operation). The loose connection caused high resistance and heat that eventually created a weakened electrical path and ultimately led to failure.

2. Pipeway and Steam Drum Cooling Units

The leak into the enclosure dirty sumps was due to the leaking pipeway coolers. Both service water pumps (KE;P) restarted when power transferred to the 46 kV transmission line. This caused a mild pressure surge that was sufficient to cause the pipeway cooler tubes (TRB) to develop leaks. After a thorough walkdown, other components within the service water system do not appear to have been damaged.

3. Radiation Area Monitor (RI-R009A) Channels (CHA) 1 Through 5

A capacitor (CAP) failed in the high voltage power supply (JX) during the transfer of station power from 138 kV to the 46 kV transmission line. The units are designed to withstand such power excursions, however in this case a weak capacitor caused unit failure.

Exceeding Differential Temperature Limit On Reactor Vessel

4. Upon loss of the 138 kV transmission line, feed pumps and reactor recirculating pumps were temporarily lost. Upon transfer to the 46 kV line both CRD pumps started and fed 50 gpm of cool water into the

Primary System (AD). A portion of this cooler water entered through the CRD seals and the rest through the regenerative heat exchangers (HX) to the lower reactor head. Flow from natural circulation was not sufficient to sweep the cooler water from the lower head, therefore a higher than desired cooldown rate on the lower head of the vessel was experienced.

CORRECTIVE ACTION TAKEN

1. States Terminal Block/Connector

Consumers Power Region Repair and Lab Service were called to investigate the substation equipment. Lab Services found the protective relays were set correctly and performed as designed. Region Repair checked and tested the main transformer, and concluded that the windings were not damaged.

Lab Services visually checked the terminations for the current transformers and 387 protective relaying. No problems were reported in the control room terminals. Terminals in the substation were discovered to be damaged from overheating due to high resistance.

Lab Services replaced the terminal strips and wires in the substation panel (PL) and successfully tested the 199 OCB. The normal station power lineup (199 OCB closed, 1126 OCB closed and 7726 OCB lined up for automatic operation) was accomplished on October 30, 1992 at 2010.

2. Pipeway and Steam Drum Cooling Units

The A and B cooling units were repaired, tested and were declared operable on November 2, 1992 at 1655 and 1738 respectively.

3. Radiation Area Monitor (RI-R009A) Channels 1 through 5

On October 29, 1992 RI-R009A was removed, and the high voltage power supply was replaced. RI-R009A was then reinstalled and calibrated. The area monitoring channels were declared operable at 1914 that same day.

4. Exceeding Differential Temperature Limit On Reactor Vessel

After an hour, a reactor recirculating pump was started, terminating the pooling of the water in the lower head by promoting mixing of the primary coolant. Temperature of the vessel rose at a rate of approximately 124 degrees F/hour, within a half hour, and an additional 20 degrees F during the following half hour.

A similar event occurred in January of 1972, and at that time it appears that data contained in Amendment 8 was used in part to satisfy concerns that vessel stress limits were not exceeded. The control rod drive nozzles (AA;NZL) were the limiting case with 8000 allowable thermal cycles.

Structural Integrity Associates Inc were contacted to evaluate the current event and its effect on the CRD nozzles and the reactor vessel. They concluded that the thermal transient associated with the trip does not represent a fatigue concern for the vessel. This evaluation was completed prior to plant startup on November 2, 1992.

ACTION TAKEN TO PREVENT RECURRENCE

The following actions are intended to prevent recurrence:

1. Soon after plant startup, an infra-red inspection of the substation electrical panels will be performed.

2. Determine what future inspections should be done in the substation panels and create a frequency for the inspections.

3. The pipeway coolers will continue to be managed in accordance with the Integrated Plan.

4. An investigation will be completed to determine if pressure surges in the service water system can be reduced in response to station power events of this type.

5. A scenario will be developed and training will be conducted on this particular event for plant operating crews.

6. Even though operator response was appropriate for this transient, an investigation will be made by the department to determine if a change in operating philosophy is warranted to prevent exceeding vessel temperature differentials during future postulated events of this nature.

SAFETY SIGNIFICANCE

This event is bounded by two higher order transients, Turbine Trip Without

Bypass and Recirculation Pump Seizure, which limit the safety significance of this event. Other contributing factors include the successful manual scram of the reactor from 98% power, the successful transfer of station power from the 138 kV transmission line to the 46 kV transmission line, and the fact that the Engineered Safety Features were never challenged.

In regards to exceeding vessel temperature differential of 150 degrees F during the event, an evaluation was performed by Structural Integrity Associates Inc before the facility was returned to power operation. They concluded that the fatigue damage produced by the event was only slightly greater than that associated with normal startup/fast shutdown transient addressed in the design stress analysis, for which the allowable number of cycles is 8000. Therefore the thermal transient associated with the trip does not represent a fatigue concern for the vessel.

Note: The allowable number of cycles for the event was conservatively estimated to be 6000, so this event was equivalent, from a fatigue standpoint, to only 1.33 of the 8000 cycles.

ATTACHMENT 1 TO 9212010278 PAGE 1 OF 1

Consumers
Power
William L Beckman
Plant Manager
POWERING
MICHIGAN'S PROGRESS

Big Rock Point Nuclear Plant, 10269 US-31 North, Charlevoix, MI 49720

November 25, 1992

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT -
LICENSEE EVENT REPORT 92-014: LOSS OF TRANSMISSION SYSTEM
RESULTING IN
MANUAL REACTOR SCRAM (RPS ACTUATION) AND ELECTRIC AND DIESEL
FIRE PUMP
START (ESF ACTUATION)

Licensee Event Report (LER) 92014, Loss of Transmission System Resulting in Manual Reactor Scram (RPS Actuation) and Electric and Diesel Fire Pump Start (ESF Actuation), is attached. This event is reportable to the

Nuclear Regulatory Commission per 10 CFR 50.73(a)(2)(iv).

William L Beckman
Plant Manager

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point

ATTACHMENT

A CMS ENERGY COMPANY

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